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Containment Integrity Research Program Plan

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

This report presents a plan for research on the question of containment performance in postulated severe accident scenarios. It focuses on the research being performed by the Structural and Seismic Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research. Summaries of the plans for this work have previously been published in the "Nuclear Power Plant Severe Accident Research Plan" (NUREG-0900). This report provides an update to reflect current status. This plan provides a summary of results to date as well as an outline of planned activities and milestones to the contemplated completion of the program in FY 1989.

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FOREWORD

This report presents a plan for research on the question of containment performance in postulated severe accident scenarios. It focuses on the research being performed by the Structural and Seismic Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research. Summaries of the plans for this work have previously been published in the "Nuclear Power Plant Severe Accident Research Plan" (NUREG-0900). This report provides an update to reflect current status. This plan provides a summary of results to date as well as an outline of planned activities and milestones to the contemplated completion of the program in FY 1989.

Research efforts on containment integrity have had, from inception, the benefit of comments and suggestions from a peer review panel composed of recognized experts from industry and universities.

Publication of this plan will make more visible to industry and other interested persons what our objectives are and how we are approaching the work in these important areas. It is intended that this plan will periodically be updated; therefore, comments on this plan are welcomed from all quarters. Comments need not be restricted to activities for the period covered but may include comments on omissions or what might be considered for the longer term. Please address comments directly to:

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1. TECHNICAL ISSUES

Risk analyses indicate that containment performance plays a dominant role in assessing of risk associated with severe accidents. A key insight emerging from the research on accident releases is that the mode and timing of containment failure are very important in determining accident consequences. Early failure without other mitigating factors can result in large radioactivity releases, while delayed failure of even several hours can significantly reduce the amount of radioactive material available for release. Hence, the ultimate concern of the containment performance issue is how well the containment, conservatively designed for a postulated loss-of-coolant accident (LOCA), can withstand the pressure and temperature associated with severe core damage accidents. For scenarios in which containment integrity is maintained, consequences are small. In those scenarios leading to containment failure, consequence predictions depend on both timing and type of failure. The manner by which it fails would influence the amount of airborne radioactive materials that could be released outside the containment. Knowledge of the time interval during which containment leaktight capability is ensured is important because, if the time interval between the release of radioactive material and containment failure is long, substantial fission product deposition will occur within containment. Furthermore, the mode in which containment fails, i.e., gross failure versus leakage through failure of penetrations, could influence the amount of radioactive material inside the containment that would be released outside the containment.

Two basic models can be used in severe accident risk estimation to characterize the loss of containment integrity: the "threshold" model and the "leakage-before-failure" model. Virtually all risk assessments performed to date have used the threshold model, which defines a threshold pressure, with some associated uncertainty range, at which the containment will fail with the potential atmosphere (which may contain a large amount of fission products). If the containment pressure loading is calculated to be below the threshold pressure range, the containment is considered to be intact and the offsite consequences are therefore quite low. This model was used in the Reactor Safety Study (Ref. 1). Some recent analyses (Ref. 2) have pointed out that gross containment failure is not the only pathway to substantial offsite consequences and that significant leakage on the order of 100 volume percent per day or more may result (if it occurs sufficiently early in the accident sequence) in substantial offsite releases of radioactive fission products. The leakage-before-failure model provides a means of accounting for this condition when performing risk assessment analyses.

There are, however, significant uncertainties associated with predictions of leakage from containments. The technical basis for containment design was intended to ensure very low leakage under postulated LOCAs. No explicit consideration was given to performance under severe accidents. The analytical methods and test procedures used for design are aimed at ensuring that leaks are prevented, not predicting the onset or growth of leakage. Consequently, attempts to introduce a more realistic leakage model into predictions of

containment performance have essentially required the development and verification of new techniques to predict severe accident loadings on and inelastic response of containment structures. Some elements of this new approach are further along in their development than others. The following sections indicate the current status of the development of tools necessary for confident prediction of containment performance under the pressure and temperature levels associated with severe accidents.

Predictions of the pressure and temperature histories inside a containment during a severe accident are the subject of current state-of-the-art research. Sources of containment loading resulting from a severe accident depend on the sequence and the progression of the accident, including phenomena such as: (For more details on these loads refer to the cited sections of Appendix J to Ref. 3.)

1. Containment pressure loading from primary system blowdown during and after core degradation, including decay heat [J.5];

2. Pressure loading from hydrogen burn [J.4];

3. Pressure loading from high-pressure melt ejection: (a) from a steam spike alone, (b) from a steam spike and direct containment heating, and (c) from metallic-chemical reaction and direct containment heating, including hydrogen phenomena [J.5];

4. Pressure from noncondensible gases generated during core-concrete interaction [J.6]; and

5. High temperatures late in the transient that could induce failure [J.2].

There is still significant uncertainty in the scientific community about the adequacy of understanding of the phenomena leading to some containment loads. Nonetheless, confident predictions of containment performance under bounding estimates of the loads can serve the useful purpose of permitting distinctions among scenarios likely to lead to early, late, or no containment failure.

Finally, there are two significant failure modes of containment that are not within the scope of this research plan but are the subject of other research programs. First, there is the possibility, in some scenarios involving melt-through of the reactor vessel, of a leak path developing as a result of core debris contacting the containment (Ref. 4). Secondly, there is the possibility of pre-existing leak paths that could lead to fission product release from the containment if an accident were to occur (Ref. 5). A diagram is shown in Figure 1 that puts the research program on containment integrity in the context of other efforts in research on severe accident consequences.

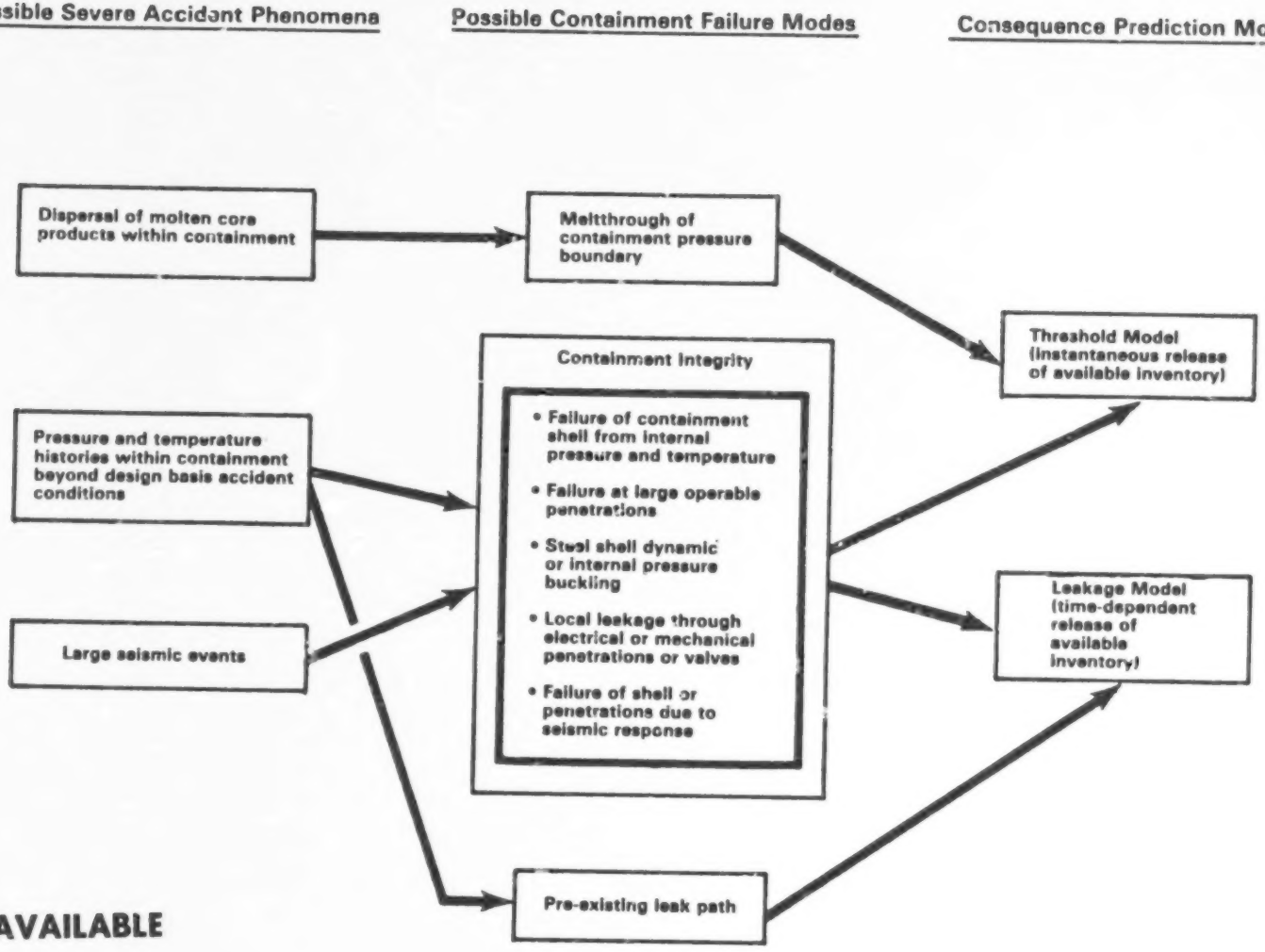
Possible Severe Accident Phenomena
Possible Containment Failure Modes
Consequence Prediction Models
Dispersal of molten core products within containment
Meltthrough of containment pressure boundary
Threshold Model (instantaneous release of available inventory)
Containment Integrity
• Failure of containment shell from internal pressure and temperature
• Failure at large operable penetrations
• Steel shell dynamic or internal pressure buckling
• Local leakage through electrical or mechanical penetrations or valves
• Failure of shell or penetrations due to seismic response
Pressure and temperature histories within containment beyond design basis accident conditions
Large seismic events
Leakage Model (time-dependent release of available inventory)
Pre-existing leak path

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Figure 1 Containment Integrity Research Element in Context of Severe Accident Research.

2. REGULATORY NEEDS AND RESEARCH OBJECTIVES

On August 8, 1985, the U.S. Nuclear Regulatory Commission issued a policy statement on severe accidents (Ref. 6). The policy statement provides criteria and procedural requirements for the licensing of new plants and sets goals and a schedule for the systematic examination of existing plants. On the basis of available information, the Commission concluded that existing plants pose no undue risk to the public and the Commission sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. However, the Commission emphasized that systematic examinations of existing plants are needed, encouraged the development of new designs that might realize safety benefits, and stated that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident and to mitigate the consequences of such an accident, should one occur. The NRC staff has developed an implementation program for the policy statement. The program will accomplish the goals of the policy statement within the schedule specified (Ref. 7).

For existing nuclear power plants, the Commission specified the formulation of a systematic approach to an examination of each plant now operating or under construction for possible severe accident vulnerabilities. The systematic approach will be developed during the 2 years following issuance of the policy statement. The examination of each plant will be performed by the licensees. Vulnerabilities identified by this process will be evaluated by the licensees in deciding whether corrective actions are needed. Any generic design changes that are identified by the NRC staff as necessary for public health and safety would be required through rulemaking.

The Severe Accident Policy Implementation Program provides for coordinated efforts to ensure the fulfillment of the policy contained in the policy statement. The implementation program incorporates several major elements. One element is to formulate an integrated, systematic approach for examining each nuclear power plant now operating or under construction for possible significant risk contributors that might be plant specific and might be missed without a systematic search. The examination will pay specific attention to containment performance in striking a balance between accident prevention and consequence mitigation.

The major source of risk to the public from the operation of nuclear power plants stems from accidents that lead to a containment failure. The regulatory concern is that the failure modes and associated load levels for containment structures cannot be predicted with any real confidence by the methods used for design. This is especially so if the contemplated failure mode is localized leakage. Both assessments of the risk posed by loads outside the design basis and estimates of the effectiveness of proposed mitigative steps require an ability to predict the way in which a containment will fail.

A final issue stems from the results of probabilistic risk analyses of nuclear power plants that have included attempts to model earthquake risk. In those analyses, the possibility of earthquakes larger than the design basis is included among the potential initiators of a severe accident. Typically, it is found that very large and unlikely, but not impossible, earthquake occurrences are among the dominant contributors to the probability of a severe accident. A coherent picture of the risk associated with those accident scenarios requires an improved understanding of how tolerant the containment pressure boundary is to earthquakes much larger than the design basis. The fundamental question is: Will containment performance in a severe accident be significantly degraded if the accident were initiated by a large earthquake?

The research objectives are:

1. To identify likely failure modes of the wide variety of containment designs under the loadings associated with the dominant severe accident scenarios.

2. To verify the ability of state-of-the-art calculational methods to predict actual failure modes and develop improved calculational techniques, if necessary.

3. To provide a basis for reducing the uncertainty in estimates of containment failure times used in consequence calculations.

4. To assess the ability of different containment designs to sustain challenges from very large earthquakes.

3. SCOPE OF RESEARCH

Research on containment failure modes is based on the observation that excessive leakage can occur, basically, from four sources:

1. Failure of the shell, either the containment shell itself in the case of steel containments, or the liner in the case of concrete containments;

2. Leakage at large penetrations as a result of the inelastic deformations and/or degradation of seals and gaskets;

3. Leakage at electrical penetrations due to degradation of materials under the high temperatures associated with accident scenarios; and

4. Leakage through valves due to pressure and temperature effects.

Summaries of results to date and planned efforts are presented in the following sections. Research related to shell failure or deformations of penetrations rests on analyses of and experiments on model tests of actual containment designs. These tests involve pressurization up to failure levels under ambient temperatures. Since seal and gasket materials are adversely affected by the temperatures associated with severe accidents, separate tests focusing on the development of leakage are performed under pressure and temperature conditions, usually at full scale. Experiments to examine the possibility of developing leakage through electrical penetration assemblies and valves also require experiments under temperature and pressure conditions at full scale.

4. RESULTS TO DATE

4.1 Containment Model Tests

Four 1/32-size steel models of three configurations were built and tested at Sandia National Laboratories in 1982 and 1983. The first configuration, of which there were two models, was termed a clean shell. Geometrically, the clean shell is a right circular cylinder with one end welded to a hemispherical steel dome and the opposite end welded to a thick base ring, which in turn is bolted to a rigid testing fixture. The diameter of the cylinder was about 43 inches (1.1 m) and had a height of 65 inches (1.65 m) including the dome. The thickness of the cylinder and dome material was about 0.045 inch (1.15 mm). The basemat was not modeled. The second configuration was a ring-stiffened containment that used the clean shell geometry with the addition of ten stiffening rings brazed to the cylinder wall. The third configuration was a penetration model that is also based on the clean shell geometry. Three penetrations are included in this model, representing two personnel locks and an equipment hatch. The containment models were pressurized incrementally with nitrogen gas. Strain and displacement data were recorded at each pressure increment. Due to the inherent dangers of pneumatic pressurization, testing was performed remotely in an isolated area. The steel containment models were instrumented with high elongation strain gauges, several displacement gauges, pressure transducers, and thermocouples to acquire data during the testing of the models. A coordinate determination system, which uses theodolites and the principles of triangulation, was also used to measure large displacements.

For the geometries investigated, stiffening rings attached to the containment wall increased the pressure at which the majority of the wall yields and increased the ultimate strength of the vessel. Treating the stiffening rings implicitly, by increasing the thickness of the cylinder wall by a volume equal to the increase in volume represented by the stiffening rings, i.e., smearing of the rings, was found to be a reasonable analytical procedure to predict ultimate capacity.

The penetrations in the penetration model did not have a thickened shell area around the penetration sleeves and no gaskets or seals were included in the penetrations. For these simplistic geometries, the presence of penetrations does not significantly decrease the capability of the containment. The tendency of ductile steels to flow plastically mitigates the effect of discontinuities such as penetrations. In the absence of severe flaws, estimating failure pressure using an equivalent plastic strain criterion does an adequate job for the simple geometries investigated with these analyses.

Finally, a 1/8-scale steel containment model was tested to determine its response to pressure levels exceeding the design basis. Extensive structural analyses of the model were performed prior to the test. A number of penetrations were present in this experimental model, including operable equipment hatches with single "O" ring seals, personnel lock representations, and

a constrained pipe. The model was built to ASME Code specifications with a design pressure of 40 psig. An extensive structural data base was generated during the high-pressure test of the 1/8-scale steel containment model, which was conducted November 15-17, 1984. Data were recorded at 21 different pressure levels up to and including 190 psig, which is 4.75 times the design pressure. The model ruptured after the pressure in the model was increased to 195 psig. No significant leakage was detected up to this point, although the measured displacements around the equipment hatch indicated that leakage was imminent. The membrane strains, which denote strains in the cylinder away from the effects of penetrations, were between 2.5 percent and 3 percent at 190 psig.

The containment shell is defined as the pressure boundary of the containment, exclusive of those parts of the boundary made up of penetrations, including equipment hatches, personnel airlocks, piping, and electrical penetration assemblies. A loss of integrity associated with the containment shell must involve a cohesive material failure, i.e., a throughwall crack or tear in the pressure boundary. If a throughwall crack or tear is developed, it is highly likely that rupture (unstable crack growth) will rapidly ensue.

Based on results of tests on scale models of steel containment buildings that were pressurized to failure, a simple strain criterion appears to be adequate for estimating the point at which a throughwall crack or tear will develop. The steel most commonly used in U.S. containments, A516 Gr 70, has exceptional ductility, and very large plastic strains may be developed prior to failure.

The following criterion for evaluating rupture follows from the test results:

Rupture will occur if, at any point on the pressure boundary, the equivalent strain exceeds the material's ultimate strain (the strain at maximum load as determined from a uniaxial tensile test). If the strain is primarily due to bending, somewhat higher strain may be tolerable (to account for crack growth through the thickness).

The above criterion should be used only with detailed three-dimensional finite-element analyses, which would need to include at least major penetrations and stiffeners. Results can be very sensitive to design details such as stiffener patterns around penetrations. Any failure criterion must be consistent with the analysis used to calculate the response measures used in the criterion. If axisymmetric analyses are used to predict rupture, the limiting value of strain used to predict rupture must be decreased to account for the effects of penetrations and stiffeners.

Many of the discontinuities in the containment shell, such as the springline and the base embedment, result in self-limiting states of stress, and, consequently, a cohesive material failure is not likely to initiate in these areas. On the other hand, large penetrations cause local strain concentrations in the shell that reduce the capacity relative to an undisturbed shell. Stiffeners tend to increase the general yield pressure, but they do not necessarily increase the ultimate strength by a like amount.

There is a large body of evidence that indicates that, if a throughwall crack or tear is developed, a steel containment shell will rupture catastrophically.

Kiefner (Ref. 8) developed empirical equations based on a large number of tests on pressurized cylinders that show that rupture will occur unless the initial throughwall crack is the result of a deep surface flaw (depth of the surface flaw must be at least one-half the wall thickness). Four of the five steel containment models tested at Sandia ruptured; in the one exception, leakage from a throughwall crack was sufficient to depressurize the model. However, in that particular model, the cylindrical wall had been severely thinned by excessive grinding adjacent to a repaired weld, which was equivalent to a deep surface flaw (Ref. 9). The formulas developed by Kiefner explain why rupture did not occur in that model. A final example is the J-integral analysis of the Sequoyah containment conducted by Greimann (Ref. 10). Greimann's analysis showed that a throughwall crack originating in the thinner wall sections near the springline would not be arrested at the thicker wall sections used at lower elevations, and he concluded that unstable crack growth (rupture) would occur.

It seems clear that steel containments are highly likely to rupture if a throughwall crack is developed at any point in the containment shell. It is assumed that initial surface flaws do not exceed 20 percent of the shell thickness.

Any penetration that is intended to be used for access to the containment is categorized as an operable penetration. In U.S. steel containments, this would include equipment hatches, personnel airlocks, escape hatches, and BWR Mark I and Mark II drywell heads. The potential for leakage between the sealing surfaces of these types of penetrations is outlined below.

In general, leakage from operable penetrations depends on both displacement and performance of the seal. Tests have shown that if metal-to-metal contact is maintained between the sealing surfaces, significant leakage does not occur regardless of the condition of the seal material (Ref. 11). Also, because the seals are typically compressed one-quarter of an inch, large separations of the sealing surfaces can arise without significant leakage if the performance of the seal is not compromised.

Relative displacements of the sealing surfaces can be categorized as either separation (relative motion perpendicular to the sealing surfaces) or sliding (relative motion in the plane of the surfaces). Tests on seal and gasket materials (Ref. 11) suggest that there are at least two ways in which performance of the seal can be compromised: (1) radiation or thermal aging can result in a loss of resiliency, and (2) high temperatures (500°F to 650°F, depending on the type of material) degrade the seal materials, that is, the materials outgas, then become dry and powdery.

In special cases, leakage may be a function of performance of the seal only or sealing surface deformation only. In personnel airlocks with inflatable

seals, the door and bulkhead are not designed to be in intimate contact, and, therefore, leakage will arise if the performance of the seal is compromised, irrespective of structural deformations. The performance of the seal may also be the only factor preventing leakage if there is significant out-of-flatness between the sealing surfaces. This is a condition that would not be detected by integrated leak rate tests; at design levels, seal performance prevents leakage. However, in BWR Mark I and Mark II plants, severe accident temperatures are expected to be sufficient to degrade seals, and out-of-flatness could result in leakage without additional separation or sliding of the sealing surfaces. In cases where sliding or separation of the sealing surfaces is very large, leakage can result even if the performance of the seal is not compromised.

4.2 Penetration and Materials Tests

Because of the different types of major penetrations that exist in LWR nuclear power plants and because of the large number of designs that exist for a given type of penetration, a comprehensive survey was conducted on 48 U.S. plants to determine these variations for all the major penetrations (Ref. 12). The survey that was performed by Argonne National Laboratory includes all containment types, materials, penetration designs, and all types of seals and gaskets. Based on these survey data, penetrations that are most susceptible to leakage were identified. To better evaluate the relative leakage potential for the different penetrations, a number of figure-of-merit analyses were made. These figure-of-merit analyses are based on the structural behavior of the penetration-containment system and on the geometry and material variations of the seals and gaskets that are used in the various penetrations. A comparative figure-of-merit analysis was performed on different types of penetration designs to come up with a list of penetrations that are most susceptible to leakage at beyond design conditions and, hence, may need to be tested to determine their leakage behavior.

Also, since the seal and gasket materials used in the major penetrations are expected to be a major cause of leakage of the penetrations under severe accident conditions, an extensive literature survey of the behavior and the capability of the different seal materials under different environments was conducted. This survey indicated that the performance and the capability of all the commonly used seal materials are affected by (1) temperature, particularly beyond 400°F, (2) aging due to radiation beyond a dose level of about 10^6-10^7 rad, (3) the combination of radiation and temperatures, and (4) the steam environment (mainly for silicone rubber). However, very little information was found in the literature about their behavior in the environments of interest in severe accidents. Hence, a plan has been put in place to test some of the more commonly used seal and gasket materials for severe accident conditions in order to understand their leakage behavior. Using the information from the survey of the plants and the survey of the seal and gasket materials, a test matrix for seals and gaskets has been developed. Some of the common geometries that will be tested are O-ring, tongue and groove, double dog-ear, double gumdrop, and inflatable seals. All these geometries are commonly used in large mechanical penetrations (e.g., equipment hatch, personnel airlock).

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The O-ring type is also commonly used in electrical penetration assemblies. Some of the materials to be tested are silicone rubber ethylene-propylene type rubbers (EPR/EPDM), Neoprene, and Viton. The test matrix for the seals and gaskets will evaluate the effects of radiation aging, temperature, linear scaling (i.e., effects of the length of the seal), cross section, rotation between seal mating surfaces, and gaps between mating surfaces. Full-size cross sections of the seals are used in these tests.

Tests were conducted in nitrogen, air, and steam atmospheres at temperatures up to 700°F and pressure at 155 psia. Both pressurized water reactor (PWR) and boiling water reactor (BWR) severe accident profiles, which are discussed in Reference 13, are enveloped by this environment. In a seal test, the initial temperature and pressure are raised to 360°F and 155 psia. These conditions are held overnight before raising the temperature to 500°, 600°, and 700°F and holding for 2 hours on successive days. Tests are terminated before 700°F is reached only if a seal fails and leaks.

Principal results obtained to date show that:

- Seal materials degrade as a function of time and temperature. Depending on the seal material and environment, degradation took the form of melting, charring, or brittleness.

- Leakage is limited or prevented when metal-to-metal contact is maintained at the sealing surface. In actual mechanical penetrations, the magnitude of leakage under these conditions may be higher because the surface finish and flatness of the sealing surfaces is probably not as good as in our test fixtures.

- The amount of squeeze has no apparent effect on the degradation of the seal. Metal-to-metal contact (30% squeeze) masks seal degradation by preventing or significantly limiting leakage.

- The degradation of silicone rubber under extreme environments is affected by its aging history. In a steam atmosphere, silicone rubber gaskets that were both radiated and thermally aged performed slightly better than unaged silicone rubber. Unaged silicone rubber gaskets typically failed to maintain a seal at about 500°F in a steam atmosphere. Aged silicone rubber failed to maintain a seal at about 550°F. In both cases squeeze was fixed. A decrease in squeeze during the test would probably result in leakage at lower temperatures.

- One-quarter-inch EPM/EPDM seals maintain a seal up to 650°F, regardless of aging history. As expected, the double dog-ear and double gumdrop EPDM unaged seals tested began leaking at a lower temperature (600°F) than the 1/4-inch seals. (Typically the physical properties of a given material are degraded as cross section increases.)

- Samples are aged in a fixture at the same percent squeeze that is used during the severe accident test. When subjected to aging, either thermal or radiation, all materials retain a permanent set. As a result, any subsequent decrease in squeeze (i.e., any increase in gap size) would result in leakage depending on the degree to which resiliency is lost.

- Silicone rubber degrades at a significantly lower temperature in a steam environment than in either an air or a nitrogen environment (500^{0}F versus 700^{0}F). No silicone rubber seals other than 1/4-inch O-rings have been tested.

- The degradation of EPM/EPDM is not affected by the atmosphere. One-quarter-inch EPM/EPDM O-rings typically did not leak at temperatures up to 650^{0}F in nitrogen, air, or steam atmospheres.

- The effect of pressure on seal degradation was not investigated, but it is believed to be insignificant. Leak rate obviously depends on pressure. Higher pressures may also be more effective in eroding seal material, thus increasing leakage area.

- Leakage typically occurs over less than 30 percent of the seal length. It is reasonable to expect that, given time, erosion of the seal material would cause the leakage area to increase.

4.3 Electrical Penetration Assembly Tests

As part of an effort to evaluate "containment integrity" under severe accident conditions, electrical penetration assemblies were singled out as having "one of the largest uncertainties associated with predicting the amount of radionuclides released" (Ref. 14). This potential led to a study of electrical penetration assemblies by Sebrell (Ref. 15) to assess the leak paths and potential for leakage through electric penetration assemblies in light-water-reactor (LWR) containment buildings.

Electrical penetration assemblies (EPAs) with the highest potential for leakage are (1) those with organic seals and gaskets, (2) those with elastomer O-rings on header plates, and (3) those designed for low-pressure capability. Regarding the availability of the EPAs, of the identified 18 suppliers, only three suppliers are active today from whom actual full-size EPAs similar to those supplied to the existing nuclear power plants could be obtained. These suppliers are D. G. O'Brien, Westinghouse, and Conax.

EPAs were tested to determine their leakage under severe accident environments typical of both PWRs and BWRs. These severe accident profiles are based on calculations performed under the Severe Accident Sequence Analysis (SASA) program, also funded by the NRC. Where such information is not available,

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other sources, including data from probabilistic risk assessment (PRA) studies of plants, were used. The test matrix was:

EPA	Plant	Severe Accident Environment
Conax	BWR - Mark I	700°F, 135 psia
Westinghouse	BWR - Mark III	400°F, 75 psia
D. G. O'Brien	PWR	360°F, 155 psia

The EPAs were aged for the equivalent of a 40-year life and exposed to radiation simulating that of a severe accident prior to the severe accident steam tests. Results are presented in Reference 12 and are summarized below.

D. G. O'Brien PWR Test

The D. G. O'Brien Severe Accident Conditions (SAC) test was completed according to the temperature and pressure accident conditions. There were no detectable leaks through the EPA during the steam pressurized portions, including none through the header plate O-ring seals, also called the aperture seals. A posttest air leak rate measurement at ambient temperature and 155 psia produced a leak on the order of 0.13 scc/sec. The electrical system on the EPA for all modules measured 1E+06 ohms to ground at 50 volts, and five out of the eight circuits were passing 0.5 amp to ground after 10 days. One-half amp is the maximum current permissible in the circuit designed for monitoring the current to ground. The posttest inspection showed all but one module to be electrically faulty, and the preliminary indication is that moisture traveled through the connector (also referred to as a plug) and grounded out between the module pin and the metal "mask" that surrounds each pin. It is this bridging with moisture or contaminants that causes the short to ground. There was posttest evidence of significant corrosion and chemical attack during this test. There was also a large amount of polysulfone expansion out from inside the plug bonnets and in some cases seizing between the plug and receptacle. It is important to recognize that this was a test that exceeded the design of the EPA. It was designed for 65 psig and 330°F. The test had a pressure as high as 155 psia and a temperature as high as about 390°F.

Westinghouse BWR Mark III Test

The Westinghouse SAC test was completed according to the temperature and pressure accident conditions. There were no detectable leaks through the EPA, including the header-plate O-ring seals, during the steam pressurized portions of the test and during the posttest air leak rate measurement at 75 psia.

The electrical performance of the EPA during the steam test was more a function of the EPA cables used rather than the EPA design. The first low resistance between an EPA cable and ground occurred in one of the ITT Suprenant thermocouple cables at 64 hours after the start of the SAC test. The resistance to ground of one conductor dropped from 3E+06 ohms at 500V to 1.5

Kohm. The insulator was probably damaged by the 500V potential of the Megohm meter because the resistance dropped so quickly and then slowly recovered to 1E+06 ohms at 500V. All the EPA thermocouple insulators failed in a similar manner; the resistance dropped suddenly from a value greater than 1E+05 ohms to less than 3E+04 ohms after an IR measurement with the Megohm meter. The use of the 50V to 500V potential to measure the insulation resistance of a thermocouple cable is a severe test, considering that the cable will only see a potential of less than 0.1V. The measurement probably overstressed the cable and accelerated the degradation.

Conax BWR Mark I Test

The Conax test, originally scheduled for a 10-day duration, had to be terminated after 8 days and 15 hours because of operational difficulties at the facility unrelated to the EPA test. All significant findings had emerged long before the test termination, and there was no need to rerun the experiment. There was no detectable leakage during or after the SAC test. The threshold for detection is 1E-05 scc/sec. At about 5 hours into the test, two EPA copper cable IRs to ground had dropped to less than 2 Kohms measured with the DMM. The remaining six cables had IRs to ground between 2E+05 and 1E+06 ohms at 100 VDC measured with the Megohm meter. By 11 hours into the test, all EPA copper cables had IRs to ground less than 1E+05 at 50 VDC measured with the Megohm meter and from 1500 to 1E+06 ohms measured with the DMM. The output of all EPA thermocouple cables agreed with the temperatures measured with the monitoring thermocouples during and after the SAC test. The inner EPA seal failed about 1 hour into the test, while the outer seals remained leaktight throughout the test.

4.4 Purge and Isolation Valve Tests

The Energy Technology Engineering Center completed elevated temperature and pressure leak testing on two 8-inch and one 24-inch butterfly valves typical of containment purge and vent valves.

The 24-inch valve passed the design basis elevated temperature leak test without leaking during or after the test. During the severe accident portion of the elevated temperature leak test, leakage was observed with the valve pressurized at 90 psig. The pressure was applied to the shaft upstream side of the disk. The packing was also observed to be leaking. The leakage started shortly after stabilization at 350°F. The pressure was increased to 120 psig, and the packing leakage increased to the point that it was necessary to lower the pressure to continue the test. The 350°F hold period with the shaft upstream was completed at 90 psig. The opposite side of the valve was then pressurized and the pressure increased to 120 psig. During a 4-hour hold period at 350°F, random leakage was observed; however, it was not constant. After the valve cooled down, ambient leak tests were performed. With pressure applied to the shaft upstream side of the disk, seat leakage averaged 38 standard cubic feet per hour (SCFH) at 50 psig and 320 SCFH at 125 psig. With

pressure applied to the opposite face of the disk, leakage was zero at all pressures.

The pressure-temperature combinations were also applied to the two 8-inch valves for shaft-side and nonshaft-side disk orientations. One of the valves experienced leakage at most of the test points considered while the second 8-inch valve experienced no leakage either at design basis conditions or at severe accident conditions for all disk orientations. For the first 8-inch valve, leakages were significantly higher with pressures on the nonshaft side than for the shaft-side orientation. Specifically, at design basis conditions (60 psig and 285°F), the leakage began at 58 cm^3/min and increased to 84 cm^3/min at 60 psig and 310°F. The leakage diminished to 32 cm^3/min when the pressure was increased to 90 psig with steady temperature of 310°F. No leakage was observed for pressures on shaft-side orientations. The post elevated temperature test showed the leakage to decrease from 52 SCFH (at 50 psig) to 36 SCFH (at 125 psig) with pressure on the nonshaft-side orientation. For the shaft-side orientation, the leakage remained nearly constant at 20 cm^3/min when the pressure was increased from 50 to 125 psig.

The capability of typical purge and vent valves to function and to prevent excessive leakage when subjected to loads resulting from severe accidents is important to the containment performance program. These results as well as test results to be obtained in FY 1987 will contribute to upgrading the model for estimating containment leakage.

4.5 Containment Buckling Tests

Two efforts on the buckling behavior possible in steel containments were completed at Los Alamos National Laboratory. The first, related to the effects on buckling loads caused by penetrations in cylindrical containments, was aimed at the adequacy of the design practice rather explicit prediction of containment failure, but did yield insights that will have applicability in detailed examinations of possible failure modes under seismic loading. The second, related to the post-buckling behavior of torispherical heads, applies directly to the performance of those steel containments with shallow domes, e.g., BWR Mark III.

The first effort was a comprehensive study to determine the effectiveness of the design of framing near penetrations for the prevention of buckling. Here, framing is defined as the stringer-like structural members that are attached to the shell and/or the reinforcing around and near a penetration where one or more circumferential stiffening rings are interrupted. The relative sizes of the penetrations studied in this work are typical of personnel airlocks and equipment hatches.

While the formation of a buckle does not necessarily mean that there will be a loss of containment, it would introduce conditions into the shell geometry and

material that could lead to failure. Therefore, in this study, formation of a buckle was the criterion used for evaluating the effectiveness of the framing.

It is well recognized that a sufficiently small penetration in a shell subjected to axial compression will not affect the buckling load because it merely represents a small imperfection, and there are usually other imperfections from fabrication or boundary conditions that have greater effects. As the size of the penetration is increased, however, its effect on the buckling capacity will become greater than the effect of other imperfections, and then it will determine the failure load. Tennyson (Ref. 16) and Starnes (Ref. 17) report studies that demonstrate this effect. The sizes of penetrations studied in this project are large enough to reduce the buckling capacity.

Addition of reinforcement around a penetration is used to increase the strength and buckling capacity, ideally, to the value of the unpenetrated shell. Penetrations in steel containments for nuclear reactors must be reinforced according to ASME Code rules (Ref. 18), which apply only to pressure loads. Code Case N-284 (Ref. 19) specifies the reinforcement required for buckling strength of steel containments, and it, in part, uses the area replacement rule (ARM) of Reference 18. The question arises as to whether the ARM is adequate for buckling loads and, in addition, whether it applies to penetrations with diameters over 10 percent of the containment diameter, which Reference 19 does not address. There have been numerous studies of this problem, and the results are not entirely consistent. A few of these references will be cited here.

Miller (Ref. 20) used Mylar models having containment-like geometries to study the effect of penetration size and the effectiveness of reinforcing on the buckling strength where the models are loaded in axial compression. Mylar remains elastic during the tests and, consequently, the models may be buckled many times with little changes in the buckling response. Miller constructed and tested several models, where unpenetrated shells had original buckling strengths of 50 percent or less of classical. After introducing penetrations and using reinforcing of the ARM type, he found that the buckling load in models with penetrations could be restored to the value without penetrations. Snell (Ref. 21) tested Lexan models of an airframe structure in axial compression without penetrations and with reinforced penetrations. He also concluded that the addition of reinforcement around a penetration could restore the buckling strength of a shell to its original value.

The problem was studied by Cervantes and Palazotto (Ref. 22) with a numerical model based on the Structural Analysis of General Shells (STAGS) program. The linear bifurcation portion of STAGS was used, and both ring-stiffened and stringer-stiffened shells were studied. The work indicated that a reinforcing frame adjacent to a penetration can increase the buckling load to or above that of a shell without a cutout. Starnes experimental work (Ref. 17) was found to agree well with their analysis.

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Tests were conducted on 6061-T6 aluminum shells having a radius-to-thickness ratio of about 670 by Almroth and Holmes (Ref. 23). The shells were tested in the unpenetrated condition, and with and without reinforcing around a penetration. The buckling strength of the virgin shells was from 40 to 63 percent of classical, and the penetrated and reinforced shells had buckling loads that were always less than the original load. Analysis with STAGS was in agreement with the experimental results.

Dove, Bennett, and Butler (Ref. 24) conducted tests on shells fabricated from A321 stainless steel in a study to evaluate the effectiveness of ARM in restoring the static buckling strength of the penetrated shell to the original value. The results showed that for the fabricated shells, that is, shells with typical geometric imperfections, the added reinforcing increased the strength; but whether or not the original strength was obtained was obscured by the effects of the imperfections. It was pointed out that the capacity reduction factor used in design would maintain the desired margin to failure. These tests were particularly important because the steel was used for the model material and capacity reduction factors typical for steel construction were obtained.

In the series of tests conducted by Baker, Bennett, and Babcock (Ref. 25), the buckling strength of six ring-stiffened shells having containment-like features was studied. The shells were made from a soft steel; two were unpenetrated and the others had one penetration each, of various sizes. Each penetration had reinforcing in accordance with ARM and framing representative of that used on existing steel containment structures. The results of the buckling tests on these shells were quite consistent and showed the following:

1. The buckling strength of the shell having the smallest penetration, one which did not interrupt any ring stiffeners, was unaffected by the penetration.

2. The buckling strengths of the shells having the larger penetrations were all reduced from the strength of the unpenetrated shells, the larger the penetration, the larger the reduction in strength.

It was also noted that the type of failure was a localized buckle, usually in a location adjacent to the ends of the vertical framing members. An analysis done by Meller and Bushnell (Ref. 26) for these models agreed well with the results, both for the collapse load and for the buckling modes. They also used STAGS as the numerical tool for performing their analyses.

It is of interest to note that the studies performed using plastic models and/or linear elastic materials show that the ARM reinforcing restores the model to the unpenetrated strength, but tests on steel models are either inconclusive (Ref. 24) or show that the ARM reinforcing plus the framing normally used around large penetrations may either not be adding to the buckling

strength or might even be decreasing it. Neither the plastic models nor the metallic models used in the references cited had a stress/strain curve representative of a typical containment steel. This factor is most likely the cause of the plastic models showing full recovery of buckling strength with the ARM and of the steel models not always supporting this conclusion.

The final effort (Ref. 27) was undertaken to determine the effects of framing near penetrations on buckling capacity for linear elastic materials. Based on the experimental work done for this investigation, several conclusions about the effect of framing on the buckling capacity of a containment having penetrations that have been reinforced according to the ARM method were stated:

1. The presence on an unframed penetration reinforced in accordance with ARM has little effect on buckling capacity as long as the primary membrane stresses remain elastic.

2. Addition of framing up to an "overframed" condition has a small effect on buckling capacity as long as the primary stresses remain elastic.

The tests showed a maximum increase in capacity of 11 percent and a maximum decrease of 4 percent, the amount being dependent on penetration size, location, and loading condition.

The results of the concomitant finite-element analysis led to these conclusions (Ref. 27):

1. The addition of framing to a model having an equipment hatch penetration caused a change in the buckling capacity of ± 15 percent. Whether the capacity increases or decreases depends on the loading condition.

2. The primary membrane stresses do not change significantly with the addition of framing.

A series of carefully planned and conducted buckling tests was recommended (Ref. 27) for which the models have containment-like features and the model material has the same stress/strain curve as typical containment steels. The tests conducted to date have not met these criteria, and there remain unanswered questions as to the buckling response of such models and for the prototype containment and as to the applicability of the codes used in the design and analysis of containment framing near large penetrations.

Lacking information from such a series of tests, it was recommended (Ref. 27) that, for the short term, the 25 percent reduction in buckling capacity, as defined in the interim criteria currently being used by the NRC for licensing decisions, continue to be used. Results of the proposed tests should give the information needed to determine whether the 25 percent reduction in

buckling capacity is needed or should be changed. The tests would also provide information for evaluation of individual framing designs and should provide information for the incorporation of framing rules into NRC licensing criteria and formal ASME design codes.

In steel containments with shallow domes, compressive stresses are developed in the region of sharpest curvature, known as the "knuckle region." The purpose of another Los Alamos National Laboratory investigation (Ref. 28) was to detail the investigation of the knuckle region buckling and failure for 4:1 (ratio of cylinder diameter to head height) torispherical shallow dome geometries typical of nuclear plant containments under internal pressure. Because the main interest of the NRC is in the reserve margin available before loss of containment would occur, all experiments conducted in this program were carried to rupture or loss of pressure-applying ability. In contrast, most experimental studies found in the literature dealing with knuckle buckling do not continue beyond or extensively discuss post-buckling behavior to rupture.

This program conducted several small-scale experiments on nominal 24-inch diameter heads in preparation for larger experiments. The initial experiments featured two heads that were joined at their cylindrical skirt regions to form small pressure vessels. This configuration was adequate for buckling initiation, but because the failures that occurred at the ring interface joining the two heads appeared to be related to the constraint there and to the method of joining the head to the ring, a new configuration was tested that featured the head welded on to an identical thickness cylindrical skirt that tapered into a standard 24-inch diameter pipe segment. Finally, the program culminated in two jointly funded tests carried out at the Chicago Bridge and Iron (CBI) research facilities in Plainfield, Illinois. One-half the funding for the large tests was provided by the NRC through Los Alamos with the remaining one-half provided by the ASME, CBI Industries, EPRI, EXXON, the Pressure Vessel Research Council of the United States and Japan, and the Tennessee Valley Authority.

The following major conclusions were drawn:

1. If the vessels were designed in accordance with the ASME Code, the design allowable pressure is below the buckling pressure by a factor of 1.6 to 2.4. The ASME design pressure is conservative with respect to knuckle buckling even for fabricated shells.

2. Clearly, the "margin-to-failure" in terms of loss of fluid for these shells is large. The ratio of failure pressure to design-allowable pressure for the 1/8-scale fabricated containment heads (the CBI tests) is 7 to 7.5. There is every reason to expect a similar ratio for a full-scale containment, which is an indication of much reserve margin in terms of loss of containment for these heads.

5. RESEARCH PLANS AND SCHEDULES

This section presents summaries of activities planned in the four major areas devoted to examining failure modes of containment structures subjected to pressure and temperature histories associated with scenarios. A milestone chart is shown in Figure 2. These efforts are aimed at the major objective of providing a basis for improved estimates of containment performance in severe accidents. Finally, an outline is provided for the research planned on the effects of very large earthquakes on containment performance. This effort is motivated by ongoing programs aimed at including the effects of external events in probabilistic risk assessments. The basic question concerning containment performance is: If a severe accident were to develop as a result of a large earthquake, rather than from an internal event, would containment performance be significantly different?

5.1 Containment Model Tests

A 1/6-scale reinforced concrete containment was designed by United Engineers and Constructors to a design pressure of 46 psig (317 kPa) in accordance with the ASME Code. The diameter of the model is 22 feet (6.7 m), with a total height of 37 feet (11.3 m). The cylinder wall is 9-3/4 inches (25 cm) thick, and the dome is 7 inches (18 cm) thick. This model has #4 (12-mm diameter) reinforcing bar for the eight layers of primary reinforcing (two layers of meridional, four layers of circumferential, and two layers of diagonal reinforcing steel). The containment also has a steel liner, which is 1/16-inch (1.6 mm) thick for the base and cylinder wall and 1/12-inch (2.1 mm) thick in the dome. Two equipment hatches, two airlocks, constrained penetrations, and several piping penetrations are also included in the containment. All materials used for constructing the model were selected such that their structural characteristics are the same or as close as possible to actual containment building materials.

The design of the containment model began in January 1985 and was completed in June 1985. To ensure that a high-quality model could be constructed, the design of the model and a series of preconstruction tests were conducted in parallel. Due to the containment size and schedule constraints, particular attention was paid to liner welding, splices for the reinforcing steel, concrete placement, and other construction and sequencing problems. No problems were found to preclude the containment model's construction.

Construction of the containment model began in August 1985 after comments and suggestions made by a peer review panel were incorporated into the design. The model's construction was completed in June 1986. During construction, over 250 strain gauges were applied to the reinforcing steel. In addition, several embedment gauges and thermocouples were also cast in the concrete wall. Postconstruction instrumentation includes over 100 displacement transducers and 350 additional strain gauges, and other instrumentation is planned to be applied to the containment model. At test time, there will be approximately

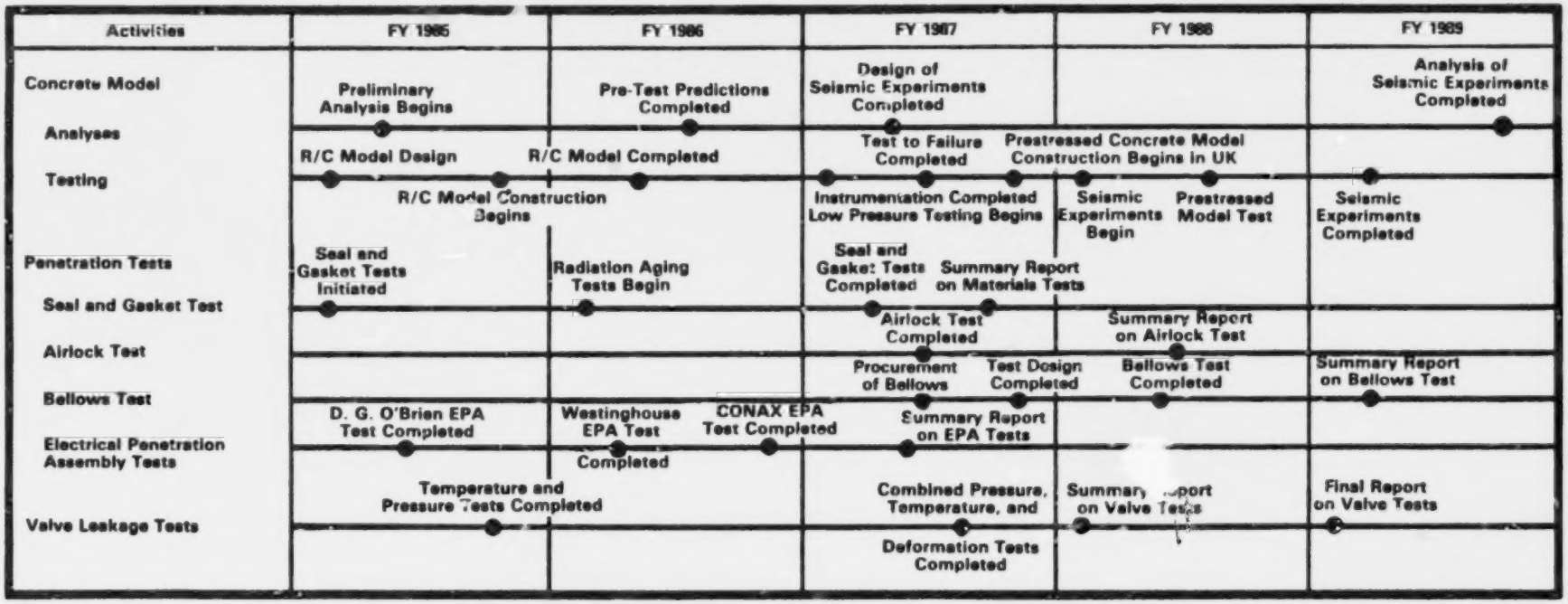

Figure 2 Planned Activities for Containment Failure Modes Research.

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1,000 channels of instrumentation. In early 1987, final checking of the data acquisition system and the overpressure testing of the containment will be performed.

Effort related to the test to failure of the 1/6-size containment model will be completed in FY 1987. Final attachment of the strain rosettes, displacement transducers, thermocouples, and RTDs (resistance temperature detectors) will also be performed before testing. Displacement transducers will be attached to measure displacements around penetrations, including deformations of the sealing surfaces, as well as general displacements of the containment shell. Temperatures about the containment wall will be measured using thermocouples. Results from these measurements will be used to correct strain gauge readings. Air temperature inside the containment will be measured using RTDs. The results from these measurements will be used for leak rate measurements.

Optical measuring and recording systems will also be installed prior to the testing of the containment model. A suite of video cameras, recorders, and monitors and still cameras will be relied upon for qualitative measurements as well as feedback for the conduct of the test.

When all data acquisition systems have been installed and checked, the model will be tested at low-pressure levels, much as a full-size containment building would be tested. While collecting structural data, a Structural Integrity Test (SIT) will be conducted--pressurizing the containment building in steps to 1.15 times its design pressure. The model will be visually inspected at its design pressure, including the mapping of substantial cracks in predetermined regions.

Following the SIT, a series of integrated leak rate tests will be conducted. The leakage of the containment model will be brought into acceptable standards (1%/day), if necessary, before further pressure testing is conducted.

Instrumentation will be added to the cracks developed during the SIT. After this has been accomplished and the data from the SIT has been digested, the containment model will be tested to failure. Failure will need to be defined during the test but will be either structural failure of a component or the containment, or excessive leakage beyond the capabilities of the pressure control system.

Pretest analyses of the 1/6-scale reinforced concrete containment model will be completed. Results will be obtained using other axisymmetric and three-dimensional models. Investigation of the possibility of local failures due to steel liner-concrete interaction will also continue. Comparisons of pretest predictions and experimental results and an assessment of capabilities for predicting the behavior of reinforced concrete containments will be completed. Because of anticipated shortcomings in the state of the art, extensive posttest analysis and code modifications may be necessary to understand differences between predictions and the experimental results.

Nine institutions (three from the U.S., three from the United Kingdom, and one each from the Federal Republic of Germany, France, and Italy) are participating in the model test.

Efforts to coordinate the activities of those participating in the analysis of the 1/6-scale model will continue. Two topical reports will be issued; the first will include a discussion of each participant's pretest predictions for the performance of the model, and the second will document comparisons with experimental data as well as any posttest analyses. Participants will be responsible for the sections in these reports that describe their findings. In addition, there will be a posttest meeting at Sandia to review the outcome of the test and a special session at the 9th Conference on Structural Mechanics in Reactor Technology that will be devoted to discussion of the testing and analysis of the model.

Efforts to extend the results of the reinforced concrete model test to permit applications to prestressed concrete containments will continue into FY 1988. The centerpiece of this effort will be a cooperative test to be performed in the United Kingdom. A 1/10-size model of the Sizewell B containment is planned for construction in 1987. The Sizewell B containment is very similar to containments of the SNUPPS design used in the U.S. A test to failure is planned for 1988. This model will provide a vehicle for testing the applicability of methods verified by the reinforced concrete model test.

The NRC research effort has proceeded with cognizance of and cooperation with an EPRI-sponsored effort on containment integrity. EPRI has sponsored one of the pretest calculations performed for the concrete containment model test and will participate in the posttest analyses of the results. In addition, EPRI has sponsored research complementary to the containment model test. EPRI's concrete containment integrity research (Ref. 29) is aimed at establishing the true failure modes and load-carrying capabilities of reinforced and prestressed concrete containments under internal pressures beyond those for which they were designed. The immediate goal is to develop a test-verified analytic tool for evaluating containment integrity. The ultimate goal, in conjunction with overall risk studies, is to characterize leak rates and radioactive releases as functions of pressure and time under various postulated severe accident scenarios.

The approach of EPRI's containment integrity research has been to conduct experimental and analytic work in parallel. To avoid the uncertainties associated with small-scale modeling of detailed structural response and leakage characterization, EPRI's experimental program features tests of large- and full-scale segments of concrete containments. Work sponsored by EPRI at the Construction Technology Laboratory of the Portland Cement Association began with simple tests on structural elements to define material behavior and progressed systematically to more prototypical tests of containment segments with penetrations and structural discontinuities to provide data on realistic liner failure mechanisms and leakage.

The analytic effort, conducted by Anatech International Corporation, has focused on verifying and applying the nonlinear finite-element code ABAQUS-EPGEN for predicting the pattern of concrete cracking and its resultant interaction with rebars and liner that could produce local failure of the liner plate.

5.2 Penetration and Materials Tests

Analytical models have been developed for both the scale representation of an equipment hatch in the 1/6-scale concrete containment and the full-size personnel airlock. Testing of both pressure-seating and pressure-unseating equipment hatches, at 1/6-scale, will be a part of the concrete containment model tests planned for mid-FY 1987. In addition, a full-scale personnel airlock, with pressure-seating seals, will be tested to failure in FY 1987 under pressure and temperature conditions. This test will be a key element in assessing the applicability of the seal and gasket materials test data for predictions of performance in actual penetrations.

Inflatable seals, aged and unaged, will be tested to pressure and temperature profiles for both PWR and BWR Mark III severe accident environments. These tests, planned for late FY 1987, seek to determine the influence of the severe accident temperature (and seal degradation) and the severe accident pressure on leakage through inflatable seals. To obtain the necessary test temperatures, steam will be used to pressurize the test chamber. Leakage will be measured by condensing the steam that leaks past the seals.

Finally, a search of literature and operating data concerning bellows connections is being conducted. Plans for testing bellows will be formulated pending completion of the literature search and collection of operating data. It is anticipated that bellows could be evaluated for the effects of temperature and loading such as torsion, internal pressure, external pressure, axial deformation, and lateral motion on stability and leakage (rupture).

5.3 Electrical Penetration Assembly Tests

Testing was completed in FY 1986. A final report will be published in FY 1987 documenting the results of the tests.

5.4 Purge and Isolation Valve Tests

Tests of containment isolation system valves (butterfly valves, small globe valves, and gate valves) under combined pressure, temperature, and loadings due to containment and displacements will be completed in FY 1987. Typical piping supports and penetrations will also be used. The purpose of these tests is to determine whether the loads transmitted to the valves through the piping will prevent the valves from isolating under conditions caused by severe accidents.

5.5 Seismic Effects on Containment Performance

Containment structures correctly designed and fabricated in accordance with Section III of the ASME Boiler and Pressure Vessel Code can be expected to perform adequately even if subjected to seismic loads far in excess of their design basis. The fundamental reason for this expectation is that the structures were proportioned and detailed so that they would respond elastically when subjected to the combined effects of a LOCA and the safe shutdown earthquake. Although different parts of a containment are most affected by seismic loads and internal pressures, the net outcome of including the effects of both loadings is to significantly increase the capacity of the containment to withstand either loading, considered separately. In addition, the design criteria specified essentially elastic response. Since significant yielding must occur before the onset of failure, that aspect of design practice adds to the margin above design levels that can be expected in any containment.

Estimates of structural capacities for a number of containment shells have, indeed, indicated large expected margins in earthquake resistance. Two studies are especially worth noting. One (Ref. 30) summarized estimates of failure levels for concrete containments made in connection with seismic PRAs. The other (Ref. 31) focused on providing estimates of the resistance of steel containment shells that could be used in probabilistic evaluations. The plants studied were all located in the Eastern U.S. (i.e., east of the Rocky Mountains) in regions of low-to-moderate seismic activity and had safe shutdown earthquakes ranging from 0.10g to 0.25g. The concrete containment study estimated median capacities ranging from 2.5g to 9.7g. The steel containment study provided mean estimates of capacity ranging from 1.45g to 16.25g. Considerable uncertainty is associated with these estimates. For example, in the concrete containment study, an attempt was made to include the effects of both randomness and uncertainty and derive capacity estimates that were felt to have a high confidence of low probability of failure (HCPLF). The net result of reflecting the uncertainties in this was to reduce the estimated capacities by a factor of about three--with HCPLF estimates of shell capacity ranging from 0.8g to 2.7g. Notwithstanding the large uncertainties, it is still apparent that structural failure of the containment shell is not a significant potential failure mode under earthquake loading.

It is more likely that, for soil sites, any containment failure would be associated with a foundation failure, leading to failure of piping. This potential failure mode has not been studied extensively. But, in one case where it has been considered, it has been found to be dominant. In the previously cited review of seismic PRA studies (Ref. 30), the median estimates for failure of the shell and for soil failure beneath its foundation slab were 2.9g and 0.9g, respectively. When uncertainties were factored into the estimating procedure, HCPLF estimates were 0.8g and 0.3g, respectively.

The fundamental question is: Will containment performance in a severe accident be significantly degraded if the accident were initiated by a large earthquake?

A soil failure would amplify motions at piping penetrations and could create a leak path. Another possible failure mode is local damage at operable penetrations, which might reduce pressure-retaining capability.

Research on containment failure modes under seismic loads will be carried out during FY 1986-1988 with the intention of assessing the ability of analytical methods to predict behavior near failure. It is important to remember that containment design technology has focused on assurance of performance under design conditions--not margins to failure. The first effort, currently in progress, is a detailed examination of six plants (three PWRs, three BWRs) to identify possible failure modes under large seismic loads. This scoping effort will be completed in the summer of 1987. The extent to which predictions of failure at the most likely locations rest on calculations that have been verified by experiment is of particular significance. There is a distinct possibility that, notwithstanding the wide differences in details for existing containment designs, it will be possible to show with confidence that implausible earthquake levels would be required to cause failure at most locations. But it is also likely that, for some designs, some failure modes will be predicted at levels of interest. Necessary experiments will be identified and planned during FY 1987-1989 and carried out in FY 1988 and 1989. It is likely that the outcome of this research will, in the context of PRAs, improve confidence in mean or median estimates of failure level.

Discussions are currently under way exploring the possibilities for cooperation with the Japan Atomic Energy Research Institute and the Ministry for International Trade and Industry on the question of containment performance under severe earthquakes. It is possible that exchanges of information about experiments and analyses conducted on this topic may be found to be mutually beneficial and will be implemented.

REFERENCES

1. U.S. Nuclear Regulatory Commission (USNRC), "Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

2. M. Silberberg et al., "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956, July 1986.

3. USNRC, "Reactor Risk Reference Document," NUREG-1150, Vol. 3, Draft for Comment, February 1987.

4. K. R. Perkins et al., "Containment Loading for Severe Accidents in BWRs with a Mark I Containment," Brookhaven National Laboratory Report, November 1984.*

5. P. J. Pelto et al., "Reliability Analysis of Containment Isolation Systems," Battelle Pacific Northwest Laboratories, NUREG/CR-4220, PNL-5432, June 1985.

6. USNRC, "Policy Statement on Severe Reactor Accidents," Federal Register, Vol. 50, p. 32138, August 8, 1985.

7. USNRC, "Implementation Plan for the Severe Accident Policy Statement and the Regulatory Uses of New Source-Term Information," December 1986.*

8. J. F. Kiefner et al., "Failure Stress Levels of Flaws in Pressurized Cylinders," Progress in Flaw Growth and Fracture Toughness Testing, ASTM STP536, American Society for Testing and Materials, pp. 461-481, 1973.

9. D. B. Clauss, D. S. Horschel, and T. E. Blejwas, "Insights into the Behavior of LWR Steel Containment Buildings During Severe Accidents," Nuclear Engineering and Design, Vol. 100, No. 2, March 1987.

10. L. Greimann, F. Fanous, and D. Bluhm, "Crack Propagation in High Strain Regions of Sequoyah Containment," Ames Laboratory, NUREG/CR-4273, Ames, Iowa, Draft, July 1985.*

11. L. N. Koenig, "Leakage Potential Through Mechanical Penetrations in a Severe Accident Environment," Proceedings of the Third Workshop on Containment Integrity (Washington, DC), NUREG/CP-0076, August 1986.

12. M. H. Shackelford et al., "Characterization of Nuclear Reactor Containment Penetration - Final Report," Sandia National Laboratories, NUREG/CR-3855, SAND84-7139, April 1985.

*Available in the NRC Public Document Room, 1717 H Street NW., Washington, DC.

13. C. V. Subramanian, "Integrity of Containment Penetrations Under Severe Accident Conditions," Proceedings of 8th SMiRT Conference, (Brussels, Belgium), Paper J 4/2, August 19-23, 1985.

14. D. H. Cook et al., "Station Blackout at Browns Ferry Unit One - Accident Sequence Analysis," Oak Ridge National Laboratory, NUREG/CR-2182, Vol. 1, ORNL/NUREG/TM-4, November 1981.

15. W. A. Sebrell, "The Potential for Containment Leak Paths Through Electrical Penetration Assemblies Under Severe Accident Conditions," Sandia National Laboratories, NUREG/CR-3234, SAND83-0538, August 1983.

16. R. C. Tennyson, "The Effects of Unreinforced Circular Cutouts on the Buckling of Circular Cylindrical Shells Under Axial Compression," Journal of Engineering for Industry, Vol. 90, pp. 541-546, November 1968.

17. J. W. Starnes, "Effect of a Circular Hole on the Buckling of Cylindrical Shells Loaded by Axial Compression," AIAA Journal, Vol. 10, No. 11, pp. 1466-1472, November 1972.

18. ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, Class MC Components, 1980 Ed.

19. ASME Boiler and Pressure Vessel Code, Code Case N-284, "Metal Containment Shell Buckling Design Methods," Section III, Division 1, Class MC, 1980.

20. C. D. Miller, "Experimental Study of the Buckling of Cylindrical Shells with Reinforced Openings," Advances in Containment Design and Analysis, M. D. Bernstein et al., Eds. (ASME), pp. 7-18, 1982.

21. R. F. Snell, "Experimental Evaluation of the Effects of Cutouts on the Stress State and Buckling Stability of the Isogrid Structures of the Thor Delta Interstage and Fairing," MDC Report G2873, April 1972.

22. J. A. Cervantes and A. N. Palazotto, "Cutout Reinforcement of Stiffened Cylindrical Shells," Journal of Aircraft, Vol. 17, No. 3, pp. 203-208, March 1979.

23. B. O. Almroth and A. M. Holmes, "Buckling of Shells with Cutouts," International Journal of Solids and Structures, Vol. 8, pp. 1057-1071, 1972.

24. R. C. Dove, J. G. Bennett, and T. A. Butler, "Buckling of Steel Cylinders Containing Circular Cutouts Reinforced According to the Area Replacement Method," Proceedings of the 1982 Joint Conference on Experimental Mechanics (Oahu-Maui, Hawaii), May 23-28, 1982.

25. W. E. Baker and J. G. Bennett, "Buckling Investigation of Ring-Stiffened Cylindrical Shells with Reinforced Openings Under Unsymmetrical Axial Loads," Los Alamos National Laboratory, NUREG/CR-3135, LA-9646-MS, March 1983.

26. E. Meller and D. Bushnell, "Buckling of Steel Containment Shells," 1-4, Lockheed Palo Alto Research Laboratory report, LMSC D812950, September 1982.

27. W. E. Baker and T. A. Butler, "A Study of the Effects of Penetration Framing on Steel Containment Buckling Capacity," Los Alamos National Laboratory, NUREG/CR-4892, LA-10977-MS, May 1987.

28. J. G. Bennett, "An Assessment of Loss-of-Containment Potential Because of Knuckle Buckling for 4:1 Steel Containment Heads," Los Alamos National Laboratory, NUREG/CR-4926, LA-10972-MS, Advance Copy.*

29. H. T. Tang and S. W. Tagart, "Containment and Piping Research," Proceedings of the Fourteenth Water Reactor Safety Information Meeting, NUREG/CP-0082, Vol. 5, pp. 523-529, February 1987.

30. R. J. Budnitz et al., "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," Appendix C, Lawrence Livermore National Laboratory, NUREG/CR-4334, UCID 20444, August 1985.

31. L. Greimann et al., "Probabilistic Seismic Resistance of Steel Containments," Iowa State University, NUREG/CR-3127, January 1984.

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This report presents a plan for research on the question of containment performance in postulated severe accident scenarios. It focuses on the research being performed by the Structural and Seismic Engineering Branch, Division of Engineering, Office of Nuclear Regulatory Research. Summaries of the plans for this work have previously been published in the "Nuclear Power Plant Severe Accident Research Plan" (NUREG-0900). This report provides an update to reflect current status. This plan provides a summary of results to date as well as an outline of planned activities and milestones to the contemplated completion of the program in FY 1989.

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